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March 24, 2003

U.S. Nuclear Regulatory Commission
Attention: Document Control Desk
Washington, D.C. 20555

Subject: Catawba Nuclear Station, Unit 1
Docket No. 50-413
Licensee Event Report 413/03-001

Attached is Licensee Event Report 413/03-001 titled "High Steam Generator Level Turbine Trip Causes Reactor Trip and Automatic Start of Motor Driven Auxiliary Feedwater System Pumps."

There are no regulatory commitments contained in this letter or its attachment.

This event is considered to be of no significance with respect to the health and safety of the public. If there are any questions on this report, please contact L.J. Rudy at (803) 831-3084.

Sincerely,

Gary R. Peterson

Attachment

IE22

xc (with attachment):

Mr. Luis A. Reyes
Regional Administrator, Region II
U.S. Nuclear Regulatory Commission
61 Forsyth Street, S.W., Suite 23T85
Atlanta, GA 30303

Mr. Robert E. Martin (addressee only)
U.S. Nuclear Regulatory Commission
Mail Stop O8-H12
11555 Rockville Pike
Rockville, MD 20852-2738

Mr. Eugene F. Guthrie
NRC Senior Resident Inspector
Catawba Nuclear Station

INPO Records Center
700 Galleria Place
Atlanta, GA 30339-5957

Marsh & McLennan, Inc.
Mr. Kenneth W. Gannaway
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Charlotte, NC 28202

NRC FORM 366 (7-2001) U.S. NUCLEAR REGULATORY COMMISSION	APPROVED BY OMB NO. 3150-0104 EXPIRES 7-31-2004
LICENSEE EVENT REPORT (LER) (See reverse for required number of digits/characters for each block)	Estimated burden per response to comply with this mandatory information collection request 50 hours Reported lessons learned are incorporated into the licensing process and fed back to industry Send comments regarding burden estimate to the Records Management Branch (T-6 E6), U S Nuclear Regulatory Commission, Washington, DC 20555-0001, or by internet e-mail to bjs1@nrc gov, and to the Desk Officer, Office of Information and Regulatory Affairs, NEOB-10202 (3150-0104), Office of Management and Budget, Washington, DC 20503 If a means used to impose information collection does not display a currently valid OMB control number, the NRC may not conduct or sponsor, and a person is not required to respond to, the information collection

1. FACILITY NAME	2. DOCKET NUMBER	3 PAGE
Catawba Nuclear Station, Unit 1	05000 413	1 OF 8

4. TITLE
High Steam Generator Level Turbine Trip Causes Reactor Trip and Automatic Start of Motor Driven Auxiliary Feedwater System Pumps

5. EVENT DATE			6. LER NUMBER			7. REPORT DATE			8. OTHER FACILITIES INVOLVED	
MO	DAY	YEAR	YEAR	SEQUENTIAL NUMBER	REV NO	MO	DAY	YEAR	FACILITY NAME	DOCKET NUMBER
02	04	2003	2003	- 001 -	00	03	24	2003	FACILITY NAME	DOCKET NUMBER

9. OPERATING MODE	1
10. POWER LEVEL	100%

11. THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR §: (Check all that apply)

	20 2201(b)		20 2203(a)(3)(ii)		50 73(a)(2)(ii)(B)		50 73(a)(2)(ix)(A)
	20 2201(d)		20 2203(a)(4)		50 73(a)(2)(iii)		50 73(a)(2)(x)
	20 2203(a)(1)		50 36(c)(1)(i)(A)	X	50 73(a)(2)(iv)(A)		73 71(a)(4)
	20 2203(a)(2)(i)		50 36(c)(1)(ii)(A)		50.73(a)(2)(v)(A)		73 71(a)(5)
	20 2203(a)(2)(ii)		50 36(c)(2)		50 73(a)(2)(v)(B)		OTHER Specify in Abstract below or in NRC Form 366A
	20.2203(a)(2)(iii)		50.46(a)(3)(ii)		50.73(a)(2)(v)(C)		
	20 2203(a)(2)(iv)		50.73(a)(2)(i)(A)		50.73(a)(2)(v)(D)		
	20 2203(a)(2)(v)		50 73(a)(2)(i)(B)		50 73(a)(2)(vii)		
	20 2203(a)(2)(vi)		50.73(a)(2)(i)(C)		50 73(a)(2)(viii)(A)		
	20 2203(a)(3)(i)		50.73(a)(2)(ii)(A)		50 73(a)(2)(viii)(B)		

12. LICENSEE CONTACT FOR THIS LER

NAME	TELEPHONE NUMBER (Include Area Code)
L.J. Rudy, Regulatory Compliance	803-831-3084

13. COMPLETE ONE LINE FOR EACH COMPONENT FAILURE DESCRIBED IN THIS REPORT

CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX	CAUSE	SYSTEM	COMPONENT	MANU-FACTURER	REPORTABLE TO EPIX

14. SUPPLEMENTAL REPORT EXPECTED				15. EXPECTED SUBMISSION DATE	MONTH	DAY	YEAR
	YES (If yes, complete EXPECTED SUBMISSION DATE).	X	NO				

16. ABSTRACT (Limit to 1400 spaces, i.e., approximately 15 single-spaced typewritten lines)

On February 4, 2003, at 1005 hours, Catawba Unit 1 tripped from 100% power. A Main Feedwater (MFW) System header pressure transmitter was being returned to service following replacement. As the transmitter was valved in, its hydraulic interaction with the other two nearby transmitters caused MFW header pressure indication to the MFW control system to fluctuate. In response to the transient, the MFW control system switched from automatic to manual control, as designed, at a pre-determined pressure difference between the transmitters. As control room operators attempted to control steam generator levels manually, a high-high level on steam generator B resulted in a turbine trip, which resulted in a reactor trip. The high-high level also resulted in a trip of the MFW pumps, which caused an automatic start of the motor driven Auxiliary Feedwater (AFW) System pumps. This event was caused by an inadequate understanding of the MFW control system response to a common impulse line hydraulic interaction. Corrective actions planned in response to this event include training appropriate plant personnel concerning MFW control system design and response and revising procedures associated with the calibration and maintenance of selected transmitters.

BACKGROUND

This event is being reported under 10 CFR 50.73(a)(2)(iv)(A), any event or condition that resulted in manual or automatic actuation of the RPS including reactor scram and reactor trip, and pressurized water reactor (PWR) auxiliary feedwater system.

Catawba Nuclear Station Unit 1 is a Westinghouse PWR [EIIS: RCT].

The purpose of the Main Feedwater (MFW) System [EIIS: SJ] is to operate in conjunction with the Condensate System [EIIS: KA] to return condensate from the condenser hotwells through the condensate polishing demineralizers and the regenerative feedwater heating cycle to the steam generators [EIIS: SG] while maintaining proper water inventories throughout the cycle. The MFW System consists of two steam turbine driven feedwater pumps [EIIS: P]; two stages of high pressure feedwater heaters [EIIS: HTR]; and piping, valves [EIIS: V], and instrumentation. Normally, both MFW pumps are in operation with each pump providing half the required feedwater flow. The feedwater passes through the high pressure feedwater heaters to a final feedwater header, where final feedwater temperature is equalized. The feedwater is then admitted to the steam generators via four feedwater lines, each of which contains a feedwater control valve and a feedwater flow nozzle. Feedwater flow to the individual steam generators is controlled by a three element feedwater control system using feedwater flow, steam generator water level, and main steam flow as control parameters for the steam generator feedwater control valves. Because of its digital design, Catawba's feedwater control system is referred to as the Digital Feedwater Control System (DFCS).

A safety function of the MFW System is to isolate the steam generators on a feedwater isolation signal. One situation in which MFW isolation is required is to prevent overfilling the steam generators should the normal means of controlling steam generator level malfunction. During an overfill situation, MFW isolation is actuated by a steam generator level high-high signal. A steam generator level high-high signal will also trip the MFW pumps and the main turbine [EIIS: TRB]. Above 69% power, a trip of the main turbine will result in a trip of the reactor.

The Auxiliary Feedwater (AFW) System [EIIS: BA] is the assured source of feedwater to the steam generators during accident conditions. The

NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

AFW System consists of two motor driven pumps and one steam turbine driven pump; various safety and non-safety related sources of AFW; and associated piping, valves, and instrumentation. There are various signals which will automatically start the AFW System. In particular, the motor driven AFW pumps are designed to start upon trip of both MFW pumps.

When this event occurred, Unit 1 was operating in Mode 1 at 100% power. Except as stated in the Event Description, no structures, systems, or components were out of service that had any significant effect on the event.

EVENT DESCRIPTION
(Certain event times are approximate)

Date/Time	Event Description
February 2, 2003/0220	Control room received an annunciator indicating trouble with DFCS. It was determined that MFW header pressure transmitter [EIIS: PT] 1CFPT5141 had failed.
February 2-3, 2003	In response to the failure of transmitter 1CFPT5141, plant personnel evaluated available options and developed a plan for replacing the transmitter with the unit on line. The plan was discussed with all affected personnel and site management. Necessary procedure changes to support the replacement were made, reviewed, and approved. A pre-job brief was held with affected personnel.
February 4, 2003/1002	DFCS was in automatic control as transmitter 1CFPT5141 was being returned to service following replacement. As the transmitter was valved in, the hydraulic interaction with nearby transmitters 1CFPT5140 and 1CFPT5142 (due to the three transmitters sharing a common impulse line) caused the MFW header pressure

NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

indication to DFCS to fluctuate. MFW pump turbine speed increased and the MFW control valves opened in response to the transient. When the difference between transmitters 1CFPT5140 and 1CFPT5142 (the controlling channel) reached 50 psi, DFCS switched from automatic to manual control, as designed.

1005 Control room operators attempted to control steam generator levels manually. High-high level on steam generator B resulted in turbine trip, which resulted in reactor trip, since the unit was above 69% power. The high-high level on steam generator B also resulted in a trip of both MFW pumps. This started the motor driven AFW pumps, as designed.

1013 Entered reactor trip response procedure and stabilized unit parameters.

1242 4 hour telephone notification made to NRC per 10 CFR 50.72(b)(2)(iv)(B) (actuation of reactor protection system while reactor is critical).

CAUSAL FACTORS

The root cause of this event was determined to be an inadequate understanding of the DFCS response to a common impulse line hydraulic interaction. The inadequate understanding was present throughout various site groups and management. This inadequate knowledge led to the development of an inadequate procedure and work plan, which upon implementation, resulted in the unanticipated MFW transient. In addition: 1) the level of risk management and oversight for the transmitter replacement activity was insufficient to ensure that the actions of a multi-disciplined team would result in event free operation; and 2) due to insufficient training, control room operator attempts to reestablish control of steam generator water levels were not effective.

CORRECTIVE ACTIONS

Immediate:

1. Operations entered the reactor trip response procedure and stabilized the unit parameters following the reactor trip.

Subsequent:

1. A team was formed to review this event and to determine its cause.

2. Procedures associated with calibration and maintenance of DFCS transmitters were placed on hold pending a review by Engineering.

Planned:

1. Training will be developed and conducted for appropriate plant personnel concerning potential interactions and consequences of common impulse line hydraulics and the design and response of the DFCS.

2. Procedures associated with calibration and maintenance of transmitters which share common impulse lines will be reviewed and revised as necessary.

3. The risk management and oversight processes used to ensure emergent work activities do not result in plant events will be strengthened to minimize the potential for events of this type.

4. Additional training will be developed and conducted for control room operators concerning response to MFW transients.

The planned corrective actions are being addressed via the Catawba Corrective Action Program. There are no NRC commitments contained in this LER.

SAFETY ANALYSIS

The Solid State Protection System functioned as designed upon receipt of the reactor trip signal. The reactor trip breakers opened within 150 milliseconds of the trip signal as designed. All control and shutdown rods inserted as designed. The rod drop time was within the Technical Specification requirement. Nuclear instrumentation response was normal following the trip.

Reactor coolant temperature control functioned as designed following the trip. In addition, primary system pressure control functioned as expected. No pressurizer power operated relief valves (PORVs) or code safety valves lifted. The pressurizer spray valves and pressurizer heaters controlled primary system pressure as designed. Pressurizer level control was normal following the trip.

Following the MFW isolation signal, all valves associated with the MFW isolation closed within the required time frame. Secondary system control functioned as designed with two exceptions. Sections of the MFW System were overpressurized in response to the MFW isolation. The peak pressure encountered on the system was 1807 psig. The design pressure on this section of the MFW System is 1385 psig; however, the observed peak pressure was less than the hydrostatic test pressure of 1.5 times the design pressure for this section of the system. Following the event, a system walkdown was performed and data trending showed no concern with the structural integrity of the MFW System due to the overpressurization transient. Also, valve 1SB15 (one of the steam dump to condenser valves) opened out of sequence and never reached its full open position as required. An air leak was found on this valve and the leak was repaired. No steam generator PORVs or main steam code safety valves lifted in response to this event. The AFW System response to this event was as designed. The motor driven AFW pumps automatically started on the loss of both MFW pumps. The steam turbine driven AFW pump did not receive a start signal, nor was it required to start in response to this event. AFW flow to all four steam generators was within the acceptable range.

In summary, the post-trip transient response remained within the bounds of the Updated Final Safety Analysis Report.

The core damage significance of this event has been evaluated quantitatively by considering the following:

- A loss of MFW initiating event.

- Actual moderator temperature coefficient value at the time of the trip.
- Actual plant configuration and maintenance activities at the time of the trip.
- Credit for recent Nuclear Service Water to AFW System modifications to minimize clam intrusion.
- Updated failure rate for Operations recovery of MFW following a loss of MFW.

The conditional core damage probability for this event is 2.0E-07, which is less than the accident sequence precursor threshold of 1.0E-06.

The dominant core damage sequences associated with this event have the significant containment safeguards systems available. These include the containment spray and hydrogen mitigation systems. Furthermore, most have low to moderate reactor coolant system pressures at reactor vessel failure. Sequences of this nature contribute insignificantly to the Large Early Release Frequency (LERF), which is dominated by the interfacing systems loss of coolant accidents and seismic initiating events. Therefore, this event is judged to have no significance with respect to the LERF for Catawba.

While this event resulted in an automatic isolation of MFW, post-trip MFW flow could have been reestablished by the control room operators if desired. Therefore, this event did not meet the initiating event criterion for a reactor trip with loss of the normal heat sink.

This event was of no significance with respect to the health and safety of the public.

ADDITIONAL INFORMATION

Within the last three years, four other reactor trip events occurred from power operation at Catawba as follows:

LER 413/00-001 described a Unit 1 reactor trip due to actuation of the turbine trip instrumentation inputs to the Solid State Protection System logic. This resulted from an electrical short within an electrical connector on the normally energized turbine

NARRATIVE *(If more space is required, use additional copies of NRC Form 366A)* (17)

electrical trip solenoid valve. The root cause of the event was a misapplication of the connector insert insulating material.

LER 414/00-003 described a Unit 2 reactor trip following a MFW transient caused by rainwater entering the turbine building and affecting control of one of the MFW pumps. The root cause of this event was determined to be inadequate oversight of a turbine building roof modification.

LER 413/01-001 described a Unit 1 reactor trip caused by a turbine trip. The root cause of this event was determined to be incomplete troubleshooting analysis associated with the main turbine protection system mechanical trip solenoid valve.

LER 414/01-003 described a Unit 2 reactor trip resulting from low reactor coolant flow when the 2D reactor coolant pump 6900 VAC feeder breaker opened in response to protective relay actuation caused by an electrical fault internal to the pump motor.

The corrective actions taken in response to these events would not have prevented this latest event from occurring. Therefore, this event was determined to be non-recurring in nature.

Energy Industry Identification System (EIIS) codes are identified in the text as [EIIS: XX]. This event is not considered reportable to the Equipment Performance and Information Exchange (EPIX) program.

This event did not involve a Safety System Functional Failure. There were no releases of radioactive materials, radiation exposures, or personnel injuries associated with this event.